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Major Modification Determination Process Utilized for Proposed Idaho National Laboratory Projects

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Abstract

Over the past three years, several new projects with the potential for major modifications to existing facilities have been considered for implementation at the Idaho National Laboratory (INL). These projects were designated to take place in existing nuclear facilities with existing documented safety analyses. 10 CFR 830.206[1] requires the contractor for a major modification to a Hazard Category 1, 2, or 3 nuclear facility to obtain Department of Energy (DOE) approval for the nuclear facility design criteria to be used for preparation of a preliminary documented safety analysis (PDSA), as well as creation and approval of the PDSA, before the contractor can procure materials or components or begin construction on the project. Given the significant effort and expense of preparation and approval of a PDSA, a major modification determination for new projects is warranted to determine if the rigorous requirements of a major modification are actually required. Furthermore, performing a major modification determination helps to ensure that important safety aspects of a project are appropriately considered prior to modification construction or equipment procurement.

The projects considered for major modification status at the INL included: treatment and packaging of unirradiated, sodium-bonded highly enriched uranium (HEU) fuel and miscellaneous casting scrap in the Materials and Fuels Complex (MFC) Fuel Manufacturing Facility (FMF); post irradiation examination of Advance Fuel Cycle Initiative (AFCI) fuel in the MFC Analytical Laboratory (AL); the Advanced Test Reactor (ATR) gas test loop (GTL); and the hydraulic shuttle irradiation system (HSIS) at ATR. The major modification determinations for three of the proposed projects resulted in a negative major modification. On the other hand, the major modification determination for the GTL project concluded that the project would require a major modification.

This paper discusses the process, methods, and considerations used by the INL for the four major modification determinations. Three of the four major modification determinations discussed herein were completed using the guidance specified in the draft of DOE-STD-1189, "Integration of Safety into the Design Process."[2] DOE-STD-1189 was released as a draft document in March 2007 and provides guidance for integrating safety considerations into the early design activities for constructing new facilities or making modifications to existing nuclear facilities. The fourth major modification determination was prepared prior to the existence of DOE-STD-1189 and was evaluated solely by the definition of a major modification given in 10 CFR 830.206. For all four projects, consideration was given to:

- Facility hazard categorization change and material inventory
- Facility footprint change with the potential to adversely affect credited safety function
- New or changed processes resulting in a change to the safety basis
- The use of new technology or equipment not approved for use in the facility
- The need for new or revised safety basis controls
- Hazards not previously evaluated in the safety basis.

1. Overview of the INL

The INL is a government-owned reservation located in southeastern Idaho (see Figure 1), approximately 25 miles west of Idaho Falls, Idaho. The INL was first established in 1949 as the National Reactor Testing Station (NTS) used for a construction and testing area for various experimental and research reactor programs, reactor fuels, structural components, materials, and reactor safety programs. The INL site covers an area of approximately 890 mi^2. The INL is currently operated by Battelle Energy Alliance, LLC (BEA) under a 10-year contract with DOE. Current missions of the INL include developing nuclear reactor technologies and supporting national security programs, advanced fuel development, spent fuel treatment, and other science and technology programs.

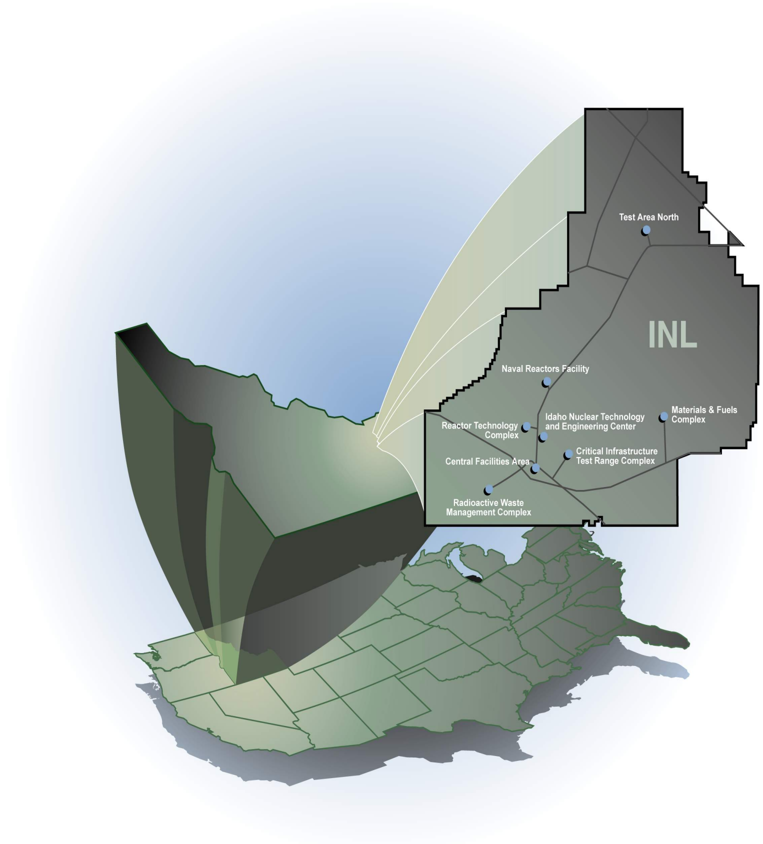

Figure 1. Location of the INL Site.

Materials and Fuels Complex

MFC is the easternmost facility located on the INL. Formerly known as Argonne National Laboratory –West and operated by the University of Chicago, the MFC site covers an area of approximately 890 acres. Construction of the MFC site began in the mid-1950s with the Experimental Breeder Reactor-II (EBR-II) and support facilities, following the successful demonstration of the EBR-I reactor which is also located on the INL reserve. The EBR-II program, which is no longer in operation, was developed for research and development of liquid metal fast breeder reactor technology. Facilities currently operated at MFC include the following:

- The Fuel Cycle Facility (FCF) is adjacent to the EBR-II facility. During EBR-II operations, this inert atmosphere hot cell facility was used as a support facility for subassembly dismantling, as well as fuel reprocessing and fuel pin casting for return to the reactor. The current mission of FCF is to process and stabilize blanket fuel from reactor programs.

- The Hot Fuel Examination Facility (HFEF) is an inert atmosphere hot cell facility. HFEF was constructed to support irradiated fuel and hardware examination programs for EBR-II and other DOE complex-wide projects. The Neutron Radiography Reactor (NRAD) is a 250 KW Training, Research, and Isotope, General Atomics (TRIGA) reactor located within HFEF. HFEF is also the home of the Waste Isolation Pilot Program (WIPP) verification project for performing visual examination of contact handled transuranic waste being shipped to the New Mexico WIPP repository.

- The Transient Reactor Test Facility (TREAT) located one mile west of the main MFC compound, is an air-cooled uranium-oxide reactor, which was used in reactor fuels and materials safety experiments using short, controlled bursts of high power nuclear energy. TREAT is currently maintained in standby pending further project identification.

- The Zero Power Physics Reactor (ZPPR) is currently in non-operational standby status. This reactor was designed for studying the properties of liquid-metal reactor cores at low power. When operational, experimental cores were built in ZPPR by hand-loading plates of reactor material into drawers. These reactor materials include uranium, plutonium, sodium, and stainless steel.

- The Laboratory and Office (L&O) Building consists of small hot cells, gloveboxes, waste-form-development equipment, and general-purpose chemistry laboratories. The AL is located within the L&O. The mission of the AL is to provide chemical, radiochemical, and physical measurements in support of MFC and INL nuclear and environmental programs.

- The Fuel Manufacturing Facility (FMF) was constructed in 1986 to house fuel manufacturing operations in support of EBR-II. Since the shut down of EBR-II, FMF has been converted to a multiuse research and development (R&D) facility.

- The Space and Security Power Source Facility (SSPSF) provides the capability for assembly and acceptance testing of radioisotope power systems (RPS) to be used in National Aeronautics and Space Administration (NASA) deep space missions and other security applications relying on an integral, secure, and long term power source.

- The Radioactive Liquid Waste Treatment Facility (RLWTF) processes low-level radioactive liquid waste generated at MFC. The facilities supported by RLWTF are EBR-II, HFEF, TREAT, ZPPR, FCF, and the AL. RLWTF is capable of evaporating approximately 227,000 L (60,000 gal) of radioactive liquid annually; the resulting residue is low-level radioactive solid waste which is packaged and stored in an environmentally acceptable form for interim-storage or shallow-land burial.

Figure 2 is an aerial view showing the major MFC facilities discussed above.

Figure 2. MFC major facilities.

Reactor Test Complex

The Reactor Test Complex (RTC) is a multipurpose experimental reactor facility complex located in the southwestern region of the INL, approximately 50 miles west of Idaho Falls. Formerly known as the Test Reactor Area, RTC was established in the early 1950s and remains a premier DOE facility supporting national nuclear technology research missions. The primary function of RTC is to maintain and operate the Advanced Test Reactor (ATR), the world's premier and largest test reactor, which is used to study the effects of irradiation of reactor fuels and structural materials. ATR also produces a rare supply of valuable medical and industrial isotopes. The reactor's unique design and high neutron flux provides an opportunity for researchers to collect data, which would normally require years to gather in a conventional reactor, in a matter of weeks or months. Historically, the primary user of ATR has been the U. S. Naval Nuclear Propulsion Program. More recently the facility has served multiple users in other government, commercial, and international missions. The reactor core is arranged in a unique clover leaf pattern, which provides nine large test spaces with additional smaller test spaces also available. The core structural components are periodically replaced in ATR, providing a consistently pristine environment for conducting high temperature, high pressure, and high flux experiments.

Other support facilities at RTC include:

- The Advanced Test Reactor Critical Facility is a low-power, full size mockup duplicate of the ATR core and provides physics data in support of ATR test programs.

- The TRA Hot Cell Facility has three shielded hot cells equipped with remote-operated tools, measuring instruments, and manipulators. The cells are used to examine highly radioactive samples from ATR and to process radioisotopes produced in ATR.

- The Radiochemistry Laboratory is used to support the Radiation Measurements Laboratory as well as to perform independent research and development projects.

- The Safety and Tritium Applications Research (STAR) Facility is used by national and international scientists performing fusion-related research and development for the DOE Office of Fusion Energy Science.

Figure 3 is an aerial view showing the major RTC facilities discussed above.

Figure 3. RTC major facilities.

2. Major Modification Requirement for Proposed Projects

The Nuclear Safety Management Rule is found in 10 CFR 830[1] and defines a major modification as "a modification to a DOE nuclear facility that is completed on or after April 9, 2001, that substantially changes the existing safety basis for the facility." Section 206 of the same part further directs that for changes to a Hazard Category (HC) 1, 2, or 3 nuclear facility meeting the definition of "major modification," the contractor must: (a) prepare a PDSA for the facility, and (b) obtain DOE approval of:

- The nuclear safety design criteria used in preparing the PDSA unless the contractor uses the design criteria in DOE Order 420.1, "Facility Safety," and
- The PDSA.

These approvals are required before the contractor can procure materials or components or begin construction. DOE may authorize the contractor to perform limited procurement and construction activities without approval of a PDSA if DOE determines that the activities are not detrimental to public health and safety and are in the best interest of DOE.

Preparation of a PDSA involves significant time and capital expenditure. Given the tight budget and schedules under which most projects operate, it is important that only those projects which truly meet the definition of a major modification are subjected to the full PDSA process. Over the past three years, four new projects being located in nuclear facilities at the INL have been evaluated for "major modification" status to determine if a PDSA was warranted. The following sections will discuss the proposed projects and the process utilized at the INL to make the final major modification determinations.

3. INL Proposed Projects

The following four subsections describe the four proposed projects that were reviewed at the INL for major modification determination. Two of the projects were proposed at RTC and two were proposed at MFC.

***Treatment and Packaging of Unirradiated, Sodium-bonded HEU Fuel*[3]**

The treatment and packaging of unirradiated, sodium-bonded HEU fuel and miscellaneous casting scrap project is aimed to process and package HEU materials currently stored at MFC's FMF for secure transfer to a designated DOE receiving facility. The specific HEU materials include approximately 7,500 unirradiated sodium-bonded EBR-II and DOE Hanford site Fast Flux Test Facility (FFTF) driver fuel elements and nearly 800 kg of HEU casting scrap from the process that formed various sodium-bonded fuels. The sodium bonding will be removed from the fuel elements in order to recover the HEU fuel and package it for off-site transport. The HEU scrap will also require repackaging in preparation for off-site transport.

Element processing begins with removing a batch size quantity of elements from storage. A batch will consist of 144 EBR-II elements or 72 FFTF elements that contain between 4.6 and 7.2 kg of U-235, depending upon the type of elements being processed. The elements are moved to the disassembly station where the spacer wire removal shears are located. This equipment is used to shear each end of the spacer wires so that it can be removed from the elements. Removal of the spacer wires is necessary so that the elements can be subsequently sectioned to remove the plenum ends. The elements are moved into an inert, argon atmosphere sodium recovery glovebox where the plenum ends are removed with a tubing cutting tool. A batch of elements is loaded into the melt-drain-evaporate (MEDE) process vessel, which is located within the glovebox. The process vessel is closed, and the process vessel pressure is reduced to approximately 200 mTorr or less. The process vessel is heated to approximately 640ºC to melt, drain, and finally evaporate the sodium from the elements. The sodium vapor is driven to a condenser where it is collected for subsequent transfer to and treatment in an MFC Resource Conservation and Recovery Act (RCRA) permitted facility. Depending upon the element type, each batch of elements will remove 100 to 450 grams of sodium. Once the sodium removal process step is completed, the process vessel is allowed to cool. The elements are removed from the process vessel, and the HEU fuel slugs are removed from the cladding. Using a shear within the glovebox, the slugs are sized as necessary for packaging.

The HEU casting scrap is currently contained within metal containers in storage. Based on a critically safe mass limit of 10 kg U-235 for a given processing batch, and assuming each current storage can contains nominally 1.4 kg of material, seven of the current storage cans would be processed at a time and during one work shift. The containers are transferred into the glovebox where they are opened and consolidated into the approved packaging for shipment.

The FMF south workroom was identified as the preferred location for this project. FMF is operated with the necessary safeguards and security controls for handling the materials. In addition, once several pieces of unused equipment are removed from the south workroom, there is ample room for the new process equipment. FMF also has an existing high-efficiency particulate air (HEPA) filter ventilation system that will be modified for this project.

Several pieces of new equipment will be installed in the designated workroom to process and package the fuel elements, the intact fuel assemblies, and the casting scrap. The new glovebox, which will contain the sodium removal process equipment, will be installed in the FMF south workroom. This glovebox will contain an inert argon atmosphere to support operations with reactive and potentially pyrophoric materials (i.e., sodium and uranium metals at elevated temperatures). A mill and support stands, which will be used in the disassembly of the fuel assemblies, will be installed. A disassembly station will be installed that will be used in the disassembly of the fuel assemblies. The disassembly station will also contain support equipment to remove the spacer wires from the elements.

Process equipment includes the necessary equipment to prepare, treat, and package the unirradiated fuel elements and assemblies such as: a mill to support disassembly of fuel assemblies; spacer wire removal shears; an element chopper to cut the elements to a size accommodated by the fuel processing baskets; MEDE equipment to evaporate, remove, condense, and collect the sodium from the fuel pins; fuel slug shears; a balance to accurately measure material for control and accountability records; and a storage container lid sealer to install a lid on the container of recovered fuel slugs.

Hazards associated with this project are releases of unirradiated U-235 from accidents, including handling events; hydrogen gas buildup from the inert atmosphere purification system; and sodium reactions associated with evaporating, draining, condensing, packaging, and storing the sodium in the fuel pins.

Advanced Fuel Cycle Initiative Post Irradiation Examination in the MFC AL[4]

This proposed activity is to develop a portion of the MFC AL basement into a shielded enclosure laboratory to be used for additional examination and analysis of advanced fuel cycle initiative (AFCI) post irradiated fuel samples. The project is funded by DOE to support the primary mission of the INL in developing fuels and materials for a new generation of commercial nuclear power plants.

As proposed, experimental fuel types will be irradiated for a period of time in the INL ATR. After the predetermined irradiation, the fuel will be removed from the reactor experiment port and transported via shielded cask to the MFC HFEF. After initial examination in the HFEF hot cells, a sub-sample will be prepared and transported by an existing pneumatic transfer system to the AL where it will be loaded into one of the shielded enclosures. Once in the shielded enclosure, the samples will be analyzed using sensitive equipment. Results of these analyses are important in the development of new reactor programs.

In order to install the post irradiation experiment (PIE) equipment and associated support equipment, various existing electrical and mechanical utilities and services require relocation. Additionally, structural modifications will be necessary to accommodate the shielding required to safely handle the samples and to operate the equipment. These modifications also include replacement of existing HEPA filters and filter housings currently in use. The enclosure inlet and outlet ventilation ducts are equipped with a HEPA filter and an automated louver with motor where the work enclosures are breached. The HEPA filters assure contamination control, and the louvers/motor assemblies control the pressure in the enclosures. In addition, this air flow control system will maintain enclosure temperatures and humidity according to the in-cell equipment operating requirements.

Process equipment includes three main pieces of analytical equipment and the support equipment for each.

The focused ion beam (FIB) detector system is a dual beam characterization instrument used to image and characterize the composition of solid materials as well as perform nano-machining of samples to prepare them for further analysis. Imaging is performed using a Schottky Field Emitter, which produces a beam of electrons that interact with the sample on a small scale. Depth of penetration of the beam into the sample depends on the material and the electron energy but is generally less than 100 nm. Interaction of the electrons with the sample material results in the production of backscattered electrons, secondary electrons, and characteristic x-rays. Detectors are situated around the FIB to detect the signals and allow for interrogation of the sample. Backscattered and secondary electron signals are used primarily for image analysis. Spectrometry of excitation released gamma rays is analyzed to determine elemental composition of the sample. The ion beam can also be used to locally remove or mill away material from the sample. This is accomplished by sputtering atoms from the sample because of the large mass collision of the ion collision compared to an electron impact.

The electron probe microanalyzer (EPMA) is an instrument used to quantitatively measure the composition of solid materials from elements with an atomic number as low as 4 (Be). The application of electric current to a filament (typically tungsten) produces a beam of electrons which interact with the sample on a small scale, typically a circle 1 to 300 μm in diameter. Depth of penetration of the beam into the sample depends on the material but is generally around 1 μm. Interaction of the electrons with the material results in the production of backscattered electrons, secondary electrons, and characteristic x-rays. Detectors are situated around the EPMA to detect these signals and allow for interrogation of the sample. Backscattered and secondary electron signals are used primarily for image analysis. X-rays pass through crystals of varying d-spacing, which allows for discrimination of specific wavelengths from the x-ray continuum. These are processed by a gas proportional detector and a matrix correction algorithm, allowing for quantitative chemical analysis of the material. Sample material can be either solid or particulate but must undergo a specific preparation procedure to use the EPMA's analytical capabilities. For a typical sample holder, the volume of the sample is limited to about 11 cm^3. Samples must be cut to size to fit, must be polished to a 1 μm finish, and must be coated if they are not conductive. Often, samples (both solids and particulate) are mounted in a metallographic mount (a hollow cylinder or ring) with epoxy, exposed to a vacuum to outgas the sample, and then polished and coated. This allows for samples far smaller than 11 cm^3 to be appropriately mounted in the sample holder so that the sample top is flat and parallel to the top of the sample holder. In addition, particulate samples prepared in this fashion become fixed in epoxy and are far safer to handle.

The micro-scale x-ray diffractometer (MXRD) generates x-rays at a source, directs them into a sample, and uses a detector to measure the intensity of the x-rays diffracted by the sample as a function of the angular relationship between the source and the detector (2Θ). In a theta-theta diffractometer, the source and detector both change angles, and the sample doesn't. The MXRD is expected to use two kinds of samples: powders and solids. When dealing with nonradioactive samples, powders are typically either glued to some kind of surface or contained in a depression in a thicker sample holder. Powders contained in depressions may be covered by x-ray transparent covers.

Hazards associated with this project are primarily release of irradiated fuel from several scenarios such as: material spill from drops; natural phenomena hazard such as earthquakes or tornados resulting in a breach of the confinement boundary or failure of shielding components; hydrogen accidents from the accumulation of hydrogen from a malfunction of the atmosphere purification system; or a major facility fire resulting in damage to confinement boundaries and subsequent release.

***ATR GTL*[5]**

The GTL is a proposed modification to the facility which will require changes to the current ATR core configuration and safety basis. Even though the proposed modifications do not involve major construction commitments, change will be required to the reactor core and subsequent changes to the safety basis. The nature of these changes were evaluated to determine if the 10 CFR 830 established criteria of "substantial" changes to the safety basis exist.

The GTL concept calls for a design which will incorporate a significant amount of "booster" fuel being added to one lobe of the ATR to gain the requisite fast neutron flux in the test region. It was determined that with the additional fuel and fissioning in that area, a significant increase in primary cooling would be required. The existing water coolant is adequate to meet the projected additional 6 to 7 MW thermal of the loop. An additional requirement will be that the lobe associated with the GTL will need to be operated at a power level of at least 40 MW in order to achieve the required fast neutron flux for the test loop. Currently, that reactor lobe is limited to a maximum power level of 34 MW per the existing ATR safety analysis report (SAR). This increase in power represents a significant step up from current levels for that lobe.

The booster fuel will be positioned in a modified version of existing flux trap baffles used to position reactor fuel elements. The flux trap baffles are identified in the ATR SAR as safety-related structures, systems, and components (SSC) required to ensure (a) the integrity of the primary coolant pressure boundary, (b) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (c) the capability to prevent or mitigate the consequences of accidents that result in potential offsite exposures comparable to the 10 CFR 100 guidelines. The ATR SAR also identifies the ATR fuel elements as safety-related equipment in that the fuel cladding serves as the first and foremost barrier to provide retention of reactor fission and transmutation products within the fuel. The GTL project proposes to add approximately 4.7 kg of U-235 booster fuel to the existing inventory in that lobe. The new fuel elements will likewise be considered safety-related, consistent with the existing fuel designation and will, therefore, be new safety-related equipment. Safety analyses are anticipated to determine the responses of new fuel and the added cooling burden in that lobe of the reactor with respect to postulated ATR accidents. The ATR operating power is measured by two systems: a thermal power calculator that measures thermal power in each of the four reactor quadrants and a lobe power calculator that is used to determine applicable operating power in each of the nine reactor flux traps or lobes. The thermal power calculation is based on measurements of the reactor coolant water as it passes through each or the four reactor core quadrants. The lobe power calculations are based on an independent system taking discrete measurements at several locations within the core. Both power measurement systems are designated as safety-related equipment. Clearly, both systems will be impacted by the significant change in fuel and power ratings. The GTL proposal, therefore, requires design changes to several reactor components or systems designated as safety-related equipment.

The GTL concept will require a sustained and long-term fast neutron flux to be maintained in the area of this test loop. The current ATR SAR limits the time of reactor operation at high powers. Changes to the SAR will be needed to justify operation at the needed levels for the GTL.

The additional fuel for the GTL booster fuel will be elements using a new fissile material form that is not currently addressed in the ATR safety basis. Supporting thermal, stress, criticality safety, and other accident analyses will be performed to provide safety basis support for the necessary changes for this project.

ATR HSIS[6]

The HSIS is being installed in the ATR to provide the capability of inserting small shuttle capsules containing target material into the reactor core. This allows the material to be irradiated and retrieved all without shutting down the reactor. This change in operating capability will result in greater reactor efficiency while reducing personnel radiation exposure. The shuttles are loaded into the system at a Send and Receive Station (SRS), located in the ATR fuel storage canal and transported to the ATR through piping installed between the canal and the reactor vessel. After irradiation, the shuttle will be transported back through the same piping to the SRS. The system is controlled by remote- and manual-operated control valves and cooled by reactor primary coolant water.

The in-core shuttle concept was previously used in ATR in the mid-1960s, including a system which was used through the 1980s but later removed to provide space for another project. At that time some of the system components were removed and never reinstalled due to lack of funding. New ATR programs and its demanding schedule will benefit from the capability provided by restoring this feature.

Construction activities include installation of transport tubing, valves, and the SRS for insertion and removal of the shuttles containing the research target material. The SRS will be installed at the west end of the canal and submerged to a depth of approximately 8 ft. Tubing will be installed from the SRS, through the canal wall, through the nozzle trench, and into the reactor vessel.

The HSIS can be separated into four major subsystems: the in-tank components, the out-of-tank components, the canal SRS station, and the shuttles themselves as follows:

1) The in-tank components include the transport and instrumentation piping from the reactor tank flange and inward, and also the in-pile portion of the system transporting the shuttle from the out-of-core piping to the irradiation position, including a position indicator to show that the shuttle has seated into the correct position.

2) The out-of-tank components include the lines connected to the reactor flange; the transport, bypass, high pressure, and instrumentation lines; and the tubing shielding where they pass through personnel-accessible areas between the reactor vessel and the SRS. Also included are isolation valves and flow meters. This system additionally will provide a means for isolating the primary coolant of the reactor vessel from the remainder of the HSIS.

3) The canal station includes a support rack, the SRS, isolation valves, and instrumentation. These components are submerged for shielding purposes.

4) The titanium shuttles are the transport vehicle for target material into and out of the reactor. In addition to transporting, the shuttle provides a primary containment boundary between the target material and the primary coolant and a confinement boundary to prevent release to the atmosphere. Each shuttle will be given a service life of a maximum 180 exposure cycles.

Potential hazards associated with the HSIS are consistent with other hazards of an operating test reactor. One hazard would result from failure of the shuttle and potential for release of irradiated target material either within the reactor core, to the transport line, or at the SRS. The SRS and sections of the transport line are located outside the containment and present the potential for an uptake exposure path to personnel and the environment. Another hazard exists from direct radiation exposure as the shuttle returns to the SRS from the reactor core, or from a "stuck" shuttle. A third operational hazard exists from the potential reactivity changes to the reactor core from the shuttle or target material as it is inserted and removed from the reactor.

These hazards are not new to the ATR operation. Drop-in capsules have long been used which will be similar to the third potential hazard condition discussed above. The other hazards will be analyzed to determine if they fall within the existing safety envelope for both risk and probability. It is recognized that additional controls may be required, such as shielding for the transport tube and a shuttle recovery plan. However, new or high-cost SSCs are not anticipated to properly mitigate these hazards.

4. DOE-STD-1189 Major Modification Evaluation Criteria

As discussed earlier, 10 CFR 830, Subpart B, "Nuclear Safety Management," dictates that a PDSA is required for major modifications to Hazard Category 1, 2, or 3 DOE nuclear facilities. In an attempt to gain more consistency throughout the DOE complex in determining what meets this subjective definition, DOE-STD-1189[2] was drafted. The draft standard includes a table, included herein as Table 1, "PDSA Evaluation Criteria." The purpose of Table 1 is to focus on the nature of the modification and the associated impact on the existing facility safety basis of the affected facility. The draft guidance for applying the table states that in applying the criteria, the intent is not to automatically trigger the need for a PDSA if one or more of the criteria are met. Rather, it is intended that each criterion be assessed individually and an integrated evaluation be performed based on the collective set of individual results. In performing this evaluation, the focus should be on the nature of the modification and its associated impact on the existing facility safety basis. Thus even a project that results in changes that ripple through the safety basis documents does not "substantially change the existing safety basis for the facility" solely because many parts or pages of the safety basis documentation need to be revised.

Table 1. DOE-STD-1189 PDSA needs evaluation.

PDSA Needs Evaluation		
Project Information		
Criterion No.	**Evaluation Criteria**	**Evaluation**
1	Add a new building or facility with a material inventory ≥ HC 3 inventory limits or increase the HC of an existing facility?	
2	Change the footprint of an existing HC 1, 2, or 3 facility with the potential to adversely impact any credited safety function?	
3	Change an existing process or add a new process resulting in a Safety Basis change requiring DOE approval?	
4	Utilize new technology or GFE not currently in use or not previously formally reviewed / approved by DOE for the affected facility?	
5	Create the need for new or revised Safety Basis controls (hardware or administrative)?	
6	Involve a hazard not previously evaluated in the DSA?	
Summary and Recommendation:		

5. Application of DOE-STD-1189 for Three INL Proposed Projects

Table 2 below shows application of the criteria from Table 1 to three of the four proposed INL projects. The GTL project major modification determination was completed prior to issuance of the draft of DOE-STD-1189, and therefore, that particular determination did not use the format found in the table. A narrative of the equivalent evaluation used for the GTL project will be discussed in the next section.

Table 2. Application of DOE-STD-1189 PDSA needs evaluation to three proposed INL projects.

PDSA Needs Evaluation				
		Evaluation		
Criterion No.	**Evaluation Criteria**	Treatment and Packaging of Unirradiated Sodium-bonded HEU Fuel	AFCI Post Irradiation Examination	ATR HSIS
1	Add a new building or facility with a material inventory >HC 3 inventory limits or increase the HC of an existing facility?	The project does not involve the addition of a new building, nor will it increase the hazard categorization of the existing facility. FMF is a HC-2 facility and will remain so with this new project.	The project does not involve the addition of a new building or facility. It will not change the hazard categorization of the facility. The AL is a HC-3 facility, and it will remain so. The projected maximum sample size is limited by the sensitivity of the equipment used for the analyses.	ATR is already classified as HC-1 nuclear facility. The addition of the HSIS will not change the hazard classification. The specific hazard of each type of material placed in the shuttles will be reviewed and approved prior to use.
2	Change the footprint of an existing HC 1, 2, or 3 facility with the potential to adversely impact any credited safety function?	The project changes the footprint of the facility in that the new glovebox will replace an existing glovebox and two chemical handling hoods. The size of the facility will not change.	The building size will not change. The basement area of the AL is currently not used as a radiological material examination area, so the usage of that area *will* change. Appropriate changes will be made in the existing safety basis documents. As an HC-3 facility, there are no SSCs which have been identified as essential to reduce hazards to acceptable levels; therefore, there is no potential to impact any credited safety function.	The actual footprint of the ATR is not changed. The significant additions with this project include a transport tube and the send receive station. The installation will impact the primary pressure boundary, the confinement, and the ATR fuel storage canal.
3	Change an existing process or add a new process resulting in a Safety Basis change requiring DOE approval?	This project will add a new process (the MEDE system) for sodium removal. The process will increase the quantity of sodium from 100 grams, currently allowed, to 433 grams based on a 72 element batch of FFTF fuel. This addition will require a safety basis change requiring DOE approval.	This project will add a process to the existing and evaluated processes in the AL. The added process is similar to the evaluated processes in that small irradiated fuel samples containing both actinide and fission products will be examined in a shielded remote-operated environment. The PIE project will not result in a safety basis change requiring DOE approval.	The installation of the HSIS will result in modification of one engineered safety feature (the confinement) and will also result in the modification of one safety related component (the canal liner). Those components will be tested for acceptable performance after the construction is complete. The description of the HSIS will be added to the ATR SAR and DOE approval obtained.

PDSA Needs Evaluation				
		Evaluation		
Criterion No.	Evaluation Criteria	Treatment and Packaging of Unirradiated Sodium-bonded HEU Fuel	AFCI Post Irradiation Examination	ATR HSIS
4	Utilize new technology or government furnished equipment (GFE) not currently in use or not previously formally reviewed / approved by DOE for the affected facility?	The project installs and utilizes MEDE equipment, which is new to the facility. The MEDE is not GFE and is not new technology. The MEDE process has been used extensively at MFC to remove the bonded-sodium from over 1,700 unirradiated EBR-II driver and blanket elements that had cladding defects. More recent tests have also been conducted with the evaporative removal of sodium from Fermi blanket fuel.	The equipment which will be installed and used in the AFCI PIE is not new technology and is not GFE. Instruments with similar technology have been approved and are in use in the affected facility.	A shuttle system was previously operated in ATR in the 1970s and 1980s. There are differences in system design from the previous system, but the basic concept remains the same: the rapid insertion and removal of material.
5	Create the need for new or revised Safety Basis controls (hardware or administrative)?	The facility fire hazards analysis and combustible loading restriction will be updated to address the MEDE process. New or revised controls will be required to limit the accumulation of recovered sodium and specify functional requirements for the packaging of recovered sodium.	New safety basis controls will not be required to perform the work under the PIE project. Existing controls are adequate for the work to be performed.	New administrative controls may be added to the ATR TSR. Other than the flange used for vessel penetration, it is not believed that any of the other new components will be classified as safety related. Changed operator actions are captured in procedures. TSR changes will be submitted to DOE for approval.
6	Involve a hazard not previously evaluated in the DSA?	The MEDE system will recover sodium batches of up to 433 grams. This exceeds the current combustible loading limit of 100 grams established for the FMF AFCI glovebox. Sodium has been addressed for FMF operations in the existing FMF DSA.	The hazards involved with examination of epoxy mounted metallographic samples of irradiated fuel sections are similar to hazards previously evaluated under the AL DSA.	Hazards associated with the project include radiation events as discussed above. These hazards are expected to be bounded by existing DSA events.

Table 2 illustrates how the various PDSA evaluation criteria questions were answered for the three projects. It should again be noted that a positive response to one or more of the evaluation criteria does not automatically trigger the need for a PDSA; rather an evaluation of the collective affect of the changes on the safety basis should be considered. As shown in the table, each of the three projects involved a new process, yet the process was determined to be similar to activities already bounded by the existing respective facility safety basis. Any new controls required were minimal. Hazards of these projects were determined to be similar to hazards included in and analyzed in existing safety basis documents. Thus even though some evaluation criteria questions were answered in the affirmative, the collective impact on the safety basis for each facility was considered minor, and each was a negative major modification determination with no PDSA required.

6. Major Modification Determination for the Proposed GTL Project

The GTL project was initiated prior to publication of the draft DOE-STD-1189 and did not follow the standardized format presented therein. However, an evaluative determination was performed on the project in a similar fashion. ATR is already classified as a HC-1 nuclear reactor facility. That classification did not change from the GTL. The facility footprint did not change; only some of the components, as discussed later, were changed. Safety functions are potentially impacted. The project represents a substantial change, including new processes from what is described in the safety basis. The technology used in the project is new to the facility and will require DOE approval for use. New controls will be needed given the number and nature of the changes from existing programs. The new fuel type and higher power levels and run times represent hazards not previously evaluated in the safety basis. The results of that determination indicate that the project would change several safety SSCs (power indicator equipment, fuel element positioners, fuel cladding), add additional safety SSCs, add significant power to one lobe of the reactor, add new and additional reactor fuel (of a different type than is currently used and discussed in the SAR), and change the reactor operations from short runs to longer, higher power and flux runs. With the additional fuel in the reactor core, a new criticality safety evaluation is needed and new thermal, stress, and accident analyses will be required. Collectively these findings indicated a substantial change in the safety basis for the facility. Thus the major modification determination was positive, and a PDSA will be required as per 10 CFR 830.

7. Summary

Several nuclear facilities at the INL are currently in the process of adding new programs to existing capabilities. The projects are: 1) Treatment and packaging of unirradiated sodium-bonded HEU fuel in FMF, 2) AFCI Post Irradiation Examination in the AL, 3) ATR GTL, and 4) ATR HSIS. Under the nuclear safety rules of 10 CFR 830, a major modification determination is required to assess the need for completing and receiving DOE approval for a PDSA prior to commencement of construction activities.

Each of these projects was evaluated independently to determine if the proposed project presented a substantial change to the respective facility's safety basis. In doing so DOE-STD-1189 Draft, or an equivalent evaluation, was used as a guide to assess the extent of the changes and the impact that those changes would have on the existing safety basis. Consideration was given to:

- Facility hazard categorization change and material inventory
- Facility footprint change with the potential to adversely affect credited safety function
- New or changed processes resulting in a change to the safety basis
- The use of new technology or equipment not approved for use in the facility
- The need for new or revised safety basis controls
- Hazards not previously evaluated in the safety basis.

Although each project answered in the affirmative to one or more of the above considerations, the overall impact on the safety basis was minimal in all projects except the ATR GTL project. Significant changes with the potential to affect the safety of the facility were identified for that project. Consequently, three of the four projects received negative major modification determinations, whereas the GTL project was given a positive determination, and a PDSA will be required. For the other three projects, revisions or addendums will be completed for the respective SARs to add description of the new processes and new controls (administrative or engineered) necessary to maintain the current safety position of the facilities.

8. References

1. 10 CFR 830, "Nuclear Safety Management," *Code of Federal Regulations,* Office of the Federal Register, January 2005.
2. DOE-STD-1189, "Integration of Safety into the Design Process," draft.
3. INL/INT-07-12417, "10 CFR 830 Major Modification Determination for Treatment and Packaging of Unirradiated Sodium Bonded HEU Fuel and Miscellaneous Casting Scrap," Idaho National Laboratory, March 2007.
4. INL/INT-07-14008, "10 CFR 830 Major Modification Determination for AFCI Fuel Post Irradiation Examination in the MFC Analytical Laboratory," Idaho National Laboratory, March 2008.
5. INEEL/EXT-05-00080, "White Paper – Advanced Test Reactor Gas Test Loop 10 CFR 830 Major Modification Position," Idaho National Laboratory, Terry A. Tomberlin, April 2005.
6. INL/INT-07-12817, "10 CFR 830 Major Modification Determination for the Hydraulic Shuttle Irradiation System at the Advanced Test Reactor (ATR)," Idaho National Laboratory, S. R. Wagoner, June 2007.